Perspectives on Severe Accident Mitigation Alternatives for U.S. Plant License Renewal

Tina Ghosh
U.S. Nuclear Regulatory Commission, Rockville, MD, USA

Robert Palla
U.S. Nuclear Regulatory Commission, Rockville, MD, USA

Donald Helton
U.S. Nuclear Regulatory Commission, Rockville, MD, USA

As part of the environmental review performed for license renewal for U.S. plants, licensees perform a severe accident mitigation alternative (SAMA) analysis. A SAMA analysis is a systematic search for potentially cost beneficial enhancements to further reduce nuclear power plant risk. This paper will provide the history of events that led to the requirement for conducting SAMA analyses and the process by which this analysis is performed. The paper will review the results from the SAMA analyses completed to date, including: (i) the onsite and offsite economic impacts of a severe accident and their typical estimated values; (ii) the types of enhancements considered/evaluated in a SAMA analysis; (iii) examples of the potentially cost-beneficial improvements (SAMAs) identified through the analyses; and (iv) the level of risk reduction that can be achieved through SAMA implementation. Finally, the paper will offer perspectives and insights on the process and results.

1. Historical context and regulatory basis

Section 5.4 of the U.S. Nuclear Regulatory Commission's (NRC), "Generic Environmental Impact Statement for License Renewal of Nuclear Plants" (NUREG-1437[1]) provides background information on the genesis of the SAMA regulatory requirement. This discussion is summarized briefly here for the purpose of providing the necessary context to the reader. Note that NUREG-1437 is in the process of being revised, but this does not affect the historical discussion provided below.

In 1980 NRC issued an interim policy statement on the consideration of severe accidents in environmental impact statements (EISs) applicable to Construction Permit and Operating License applications submitted on or after July 1, 1980[2]. The policy statement states that it is "the intent of the Commission that the staff take steps to identify additional cases that might warrant early consideration of either additional features or other actions which would prevent or mitigate the consequences of severe accidents." These features have become known as severe accident mitigation design

[1] NUREG-1437, Volume 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," U.S. Nuclear Regulatory Commission, May, 1996.
[2] 45 *Federal Register* 40101, Statement of Interim Policy, "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969," June 13th, 1980.

alternatives (SAMDAs) when applied at the design stage, or SAMAs when applied in the context of extending an existing license. But the scope of the analyses is the same.

In August 1985, NRC issued its policy statement on severe reactor accidents. That policy statement presented NRC's conclusions that existing plants pose no undue risk to public health and safety and that there was no present basis for immediate action on generic rulemaking or other regulatory changes for those plants because of severe accident risk. Nevertheless, it called for each licensee to perform an analysis designed to discover instances of particular vulnerability to core melt or unusually poor containment performance given a core-melt accident. NRC believed that this policy statement was a sufficient basis for not requiring a consideration of SAMDAs at the operating license review stage for previously constructed plants. However, a 1989 court decision ruled that such a policy statement was not sufficient to preclude a consideration of SAMDAs and that such a consideration is required for plant operation[3].

Relative to the evaluation of potential improvements for existing reactors in the U.S., the NRC gained considerable experience during the 1980s and 1990s via (a) staff assessments of SAMDAs for the Limerick, Comanche Peak, and Watts Bar plants performed as a result of the aforementioned *Limerick Ecology Action* court decision, (b) the containment performance improvement program[4], (c) the individual plant examination (IPE) program[5], and (d) the implementation of severe accident management programs at all nuclear power plants as part of an industry initiative. These regulatory programs and initiatives provide assurance that any major vulnerabilities to severe accidents have been identified and addressed, and that the residual level of risk is low. As a result, major plant modifications would not be expected as a result of a SAMA analysis. As stated in NUREG-1437, "the NRC expects that a site-specific consideration of severe accident mitigation for license renewal will only identify procedural and programmatic improvements (and perhaps minor hardware changes) as being cost-beneficial in reducing severe accident risk or consequence." This expectation has generally been met as discussed below.

2. Definition and scope

As described above, the term SAMA refers to an additional feature or action which would prevent or mitigate the consequences of serious accidents. SAMA analysis includes consideration of (i) hardware modifications, procedure changes, and training program improvements, (ii) SAMAs that would prevent core damage as well as SAMAs that would mitigate severe accident consequences, and (iii) the full scope of potential accidents (meaning both internal and external events).

---

[3] *Limerick Ecology Action v. NRC*, 869 F.d 719 (3rd Cir. 1989)

[4] NRC examined each of five U.S. reactor containment types (BWR Mark I, II and III; PWR Ice Condenser; and PWR Dry) with the purpose of examining the potential failure modes, potential enhancements, and the cost benefit of such enhancements. This examination has been called the containment performance improvement (CPI) program and was documented in a series of reports (NUREG/CR-5225; NUREG/CR-5278; NUREG/CR-5528; NUREG/ CR-5529; NUREG/CR-5565; NUREG/CR-5567; NUREG/CR-5575; NUREG/CR-5586; NUREG/CR-5589; NUREG/CR-5602; NUREG/CR-5623; NUREG/ CR-5630).

[5] In accordance with NRC's policy statement on severe accidents, each U.S. licensee was requested to perform an individual plant examination (IPE) to look for vulnerabilities to both internal and external initiating events (Generic Letter 88-20, Supplements 1-4). These examinations consider potential improvements on a plant-specific basis. Results are described in NUREG-1560 and NUREG-1742, respectively.

3. Major steps in a SAMA evaluation

3.1 Identification and characterization of leading contributors to risk

The first step of a SAMA evaluation is to identify and characterize the leading contributors to core damage frequency (CDF) and offsite risk based on a plant-specific risk study or applicable studies for other plants. In practice, maximum use is made of the plant-specific risk model for characterizing the dominant contributors to risk and identifying candidate SAMAs to address these contributors. The contribution of external events is considered to the extent that it can be supported by available risk methods, because external events can affect whether or not a SAMA is cost-beneficial (greater reduction of risk). In some cases, the SAMA may specifically relate only to external events (e.g., a modification related to a piece of hardware that is only damaged during seismic events). In other cases, a SAMA that may have been identified based on internal event considerations (e.g., use of portable generators to power equipment in a station blackout (SBO)) may also have benefits in externally initiated events (e.g., a seismic induced SBO).

3.2 Identification of candidate SAMAs

The next step in the process is to identify candidate SAMAs. Although the greatest level of risk reduction might be achieved by a major plant modification, lower cost alternatives might eliminate a substantial fraction of the risk and have a greater net benefit. In identifying SAMAs, the lowest cost means of achieving the functional objectives should not be overlooked. As an example, developing procedures to connect hydrogen igniters to portable on-site generators, rather than installing additional igniters with dedicated batteries, would be more cost-beneficial if it achieved the same reduction in risk. One key tool used in identifying SAMAs is the use of PRA importance measures (e.g., Risk Achievement Worth, or RAW) to identify important basic events from the PRA (e.g., equipment failures and operator actions) and candidate SAMAs to address these basic events. In addition, a list of SAMAs that have been found to be cost-beneficial at other plants in the past should be reviewed to identify candidate SAMAs for the plant being analyzed.

3.3 Estimation of risk reduction and implementation cost estimates

Once candidate SAMAs have been identified, an initial screening is performed to determine which SAMAs can not be cost-beneficial. A rough implementation cost estimate is developed for each SAMA. If the cost estimate exceeds the bounding condition of the maximum attainable benefit (i.e., the benefit of eliminating all plant risk) then the SAMA is screened out from further consideration because it cannot be cost-beneficial. In addition, candidate SAMAs from other plants that are not applicable to the plant being analyzed (e.g., due to design or risk-profile differences) are screened out.

For each SAMA that survives this initial screening, a benefit assessment is performed to address how the change would affect relevant risk measures (core damage frequency, offsite population dose in person-Sv [person-rem], offsite economic cost risk - OECR). This includes a description of how the change was implemented/credited in the PRA model (i.e., what changes were made to the basic events, fault trees, or event trees). For example, the impact of a procedural change might be estimated by reducing the associated human error probabilities. In some cases, bounding assumptions are used that capture the maximum possible benefit of the change, such as assuming that improvements to assure reactor cavity flooding would eliminate all containment failures due to core-concrete interactions.

A cost assessment is also performed for each SAMA. Cost estimates for hardware modifications can be taken from past studies performed for a similar plant, or developed on a plant-specific basis. Cost estimates are generally conservative in that they neglect certain cost factors (e.g., surveillance/maintenance, the cost of replacement power during implementation), therefore tending to increase the number of potentially cost beneficial SAMAs. Typically screening estimates are used for initial assessments and refined as appropriate if a SAMA is potentially cost-beneficial. In general, hardware costs are several hundred thousand to a million dollars; procedure changes range from ~$50K to $200K for complex changes with analysis and operator training impacts.

The licensee is expected to assess the impact of major uncertainties on the results, to demonstrate the robustness of the conclusions. Sensitivity analyses are typically performed, examples of which include: (1) the estimated benefits are increased by the ratio of the 95th percentile CDF to mean CDF (to address uncertainty in the CDF analysis) and (2) alternative discount rates are used in the cost-benefit analysis (e.g., 7% versus 3%) to assess sensitivity of results to the assumed discount rate.

3.4 Identification of SAMAs that are potentially cost-beneficial

To identify SAMAs that may be cost-beneficial, the net value of each SAMA is estimated. The NRC maintains two documents that provide guidance in this area: NUREG/BR-0058[6] and NUREG/BR-0184[7].

The net value of a particular SAMA can be generated from the following basic equation:

$$\text{Net Value} = (\text{APE} + \text{AOC} + \text{AOE} + \text{AOSC}) - \text{COE}$$

where:

APE = averted public exposure costs
AOC = averted offsite property damage costs
AOE = averted occupational exposure costs
AOSC = averted onsite costs = averted cleanup and decontamination costs (ACC) + averted replacement power costs (ARPC)
COE = cost of enhancement

Table 1 provides information on each of the averted cost components (offsite and onsite economic components of the maximum attainable benefit [MAB]), including references to the relevant sections of NUREG/BR-0184, the relevant supporting parameters, and aggregated values from the licensee submittals for all approved U.S. license renewals as of August 2009. The costs represent the dollar value of completely eliminating all internal event risk[8]. The MAB cost factors and total can vary widely from plant to plant due to differences in baseline risk (e.g., baseline CDF), and differences in population and land values surrounding the plant site.

[6] NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," U.S. Nuclear Regulatory Commission, September 2004.
[7] NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," U.S. Nuclear Regulatory Commission, January 1997.
[8] With the exception of data from a few licensees whose dollar values include eliminating external event risk as well.

Table 1. Supporting Information for Averted Cost Components

Cost Factor	Significance	NUREG/ BR-0184 Section	Related Parameters	Average (and Ranges) of Maximum Attainable Benefits from Licensee Submittals for All Approved License Renewals
APE	Offsite exposure	5.7.1	Δperson-Sv [Δperson-rem] (from the Level 3 PRA analysis)	\$370K (\$12K – \$1,500K)
AOC	Offsite economic	5.7.5	ΔOffsite Economic Cost (from Level 3 PRA) and accident frequency (from Level 2 PRA)	\$400K (\$10K – \$2,700K)
AOE	Onsite exposure	5.7.3	Immediate occupational dose (33 person-Sv [3,300 person-rem][a]) Long term occupational dose (200 person-Sv [20,000 person-rem][a])	\$17K (\$1K – \$130K)
ACC	Onsite economic	5.7.6.1	Onsite cleanup and decontamination cost (\$$1.1 \cdot 10^9$ single event[a], present worth)	\$870K (\$37K – \$6,300K)
ARPC	Onsite economic	5.7.6.2	Plant power level	
Total[b]				\$1,700K (\$110K - \$8,700K)

[a] From NUREG/BR-0184

[b] The range for total costs represents the range of total costs cited in the licensee submittals, not a summation of the ranges for the individual components.

3.5 More detailed analysis for remaining SAMAs

The final step in the process is a more detailed analysis of the SAMAs that were identified as being potentially cost-beneficial in the steps above. This may include a more detailed (i.e., more realistic and less bounding) evaluation of the potential benefits of the SAMA (i.e., rather than assuming that the SAMA eliminates all CDF contributors, only those sequences relevant to the SAMA are included). It may also include a more detailed development of the cost associated with the proposed modification (including such things as engineering support, training, hardware costs, and implementation costs). Additional guidance for conducting this step is available in a Nuclear Energy Institute (NEI) document NEI-05-01, Revision A[9]. The NRC staff has recommended that applicants for license renewal follow the guidance provided in NEI-05-01, Revision A, in the staff's Final License Renewal Interim Staff Guidance LR-ISG-2006-03[10].

[9] NEI-05-01 [Rev. A], "Severe Accident Mitigation Alternatives (SAMA) Analysis: Guidance Document," Nuclear Energy Institute, November 2005.

[10] LR-ISG-2006-03, "Final License Renewal Interim Staff Guidance LR-ISG-2006-03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses," U.S. Nuclear Regulatory Commission, August 2, 2007.

4. Current Status of SAMA Reviews (as of August 2009)

SAMA/SAMDA evaluations have been completed for initial plant licensing of the following three operating plants[11]: (1) Limerick (1989); (2) Comanche Peak (1989); and (3) Watts Bar 1 (1995). SAMDA evaluations have been completed for the following advanced light-water reactor certified plant designs: (1) CE System 80+ (1995); (2) General Electric Advanced Boiling Water Reactor – ABWR (1995); (3) Westinghouse Advanced Passive 600MW – AP600 (1999); and (4) Westinghouse Advanced Passive 1000MW – AP1000 (2004). To date, SAMA evaluations have been completed for operating plant license renewal applications that were approved for over 30 sites encompassing over 50 units. Table 2 lists the completed SAMA evaluations by plant, nuclear steam supply system (NSSS), and containment type.

Table 2. Completed SAMA Evaluations for Plants with Approved License Renewal

Plant Type	NSSS	Containment Type	Plant Name	Year of License Renewal Approval
BWR	GE 2	Mark I	Nine Mile Point 1	2006
			Oyster Creek	2009
	GE 3	Mark I	Dresden 2 & 3	2004
			Quad Cities 1 & 2	2004
			Monticello	2006
	GE 4	Mark I	Edwin I. Hatch 1 & 2	2002
			Peach Bottom 2 & 3	2003
			Browns Ferry 1, 2 & 3	2006
			Brunswick 1 & 2	2006
			James A. FitzPatrick	2008
	GE 5	Mark II	Nine Mile Point 2	2006

[11] NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Main Report and Supplements available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1437/.

Plant Type	NSSS	Containment Type	Plant Name	Year of License Renewal Approval
PWR	W 2-Loop	Dry Ambient	R.E. Ginna	2004
			Point Beach 1 & 2	2005
	W 3-Loop	Dry Ambient	Turkey Point 3 & 4	2002
			H.B. Robinson 2	2004
			V.C. Summer	2004
			Joseph M. Farley 1 & 2	2005
			Shearon Harris 1	2008
		Dry Subatmospheric	Surry 1 & 2	2003
			North Anna 1 & 2	2003
	W 4-Loop	Dry Ambient	Wolf Creek 1	2008
			Vogtle 1 & 2	2009
		Dry Subatmospheric	Millstone 3	2005
		Ice Condenser	McGuire 1 & 2	2003
			Catawba 1 & 2	2003
			D.C. Cook 1 & 2	2005
	CE	Dry Ambient	Calvert Cliffs 1 & 2	2000
			St. Lucie 1 & 2	2003
			Fort Calhoun	2003
			Arkansas Nuclear One 2	2005
			Millstone 2	2005
			Palisades	2007
	B&W	Dry Ambient	Oconee 1, 2 & 3	2000
			Arkansas Nuclear One 1	2001

B&W: Babcock and Wilcox
CE: Combustion Engineering
PWR: Pressurized-Water Reactor
BWR: Boiling-Water Reactor
GE: General Electric
W: Westinghouse

5. Insights from SAMA Evaluations

In general, the estimated CDFs for operating plants are relatively low (i.e., less than 10^{-4} per year). In addition, many of the weaknesses uncovered through the IPE and individual plant examination of external events (IPEEE) programs have already been addressed. It is therefore difficult to identify additional changes that both reduce risk substantially and are cost-beneficial, for the above reasons and because: (1) risk is generally driven by multiple sequences while a SAMA generally acts on only one contributor; (2) risk reduction potential is highest at operating plants (versus new reactors still under design), but the cost of implementing design changes within an operating plant is much higher too; and (3) the cost of design changes are lower in advanced light-water reactors given that the plant has not yet been constructed, but the calculated residual risk is so low that even complete elimination of all severe accident risk, if it were possible, would not warrant spending substantial funds.

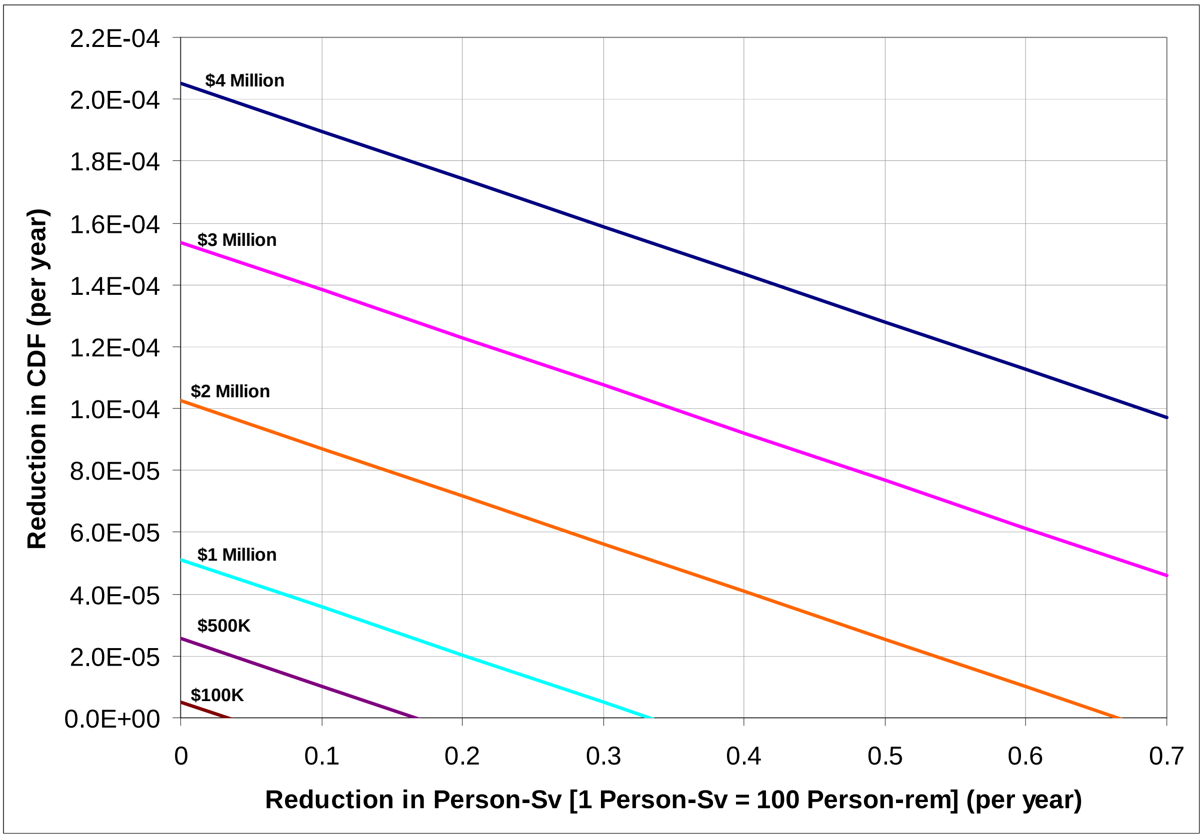
Identification of cost-beneficial changes is most likely for operating plants, where reductions in CDF could be on the order of 10^{-5} per year. At these plants, reduction in averted onsite costs and offiste impacts could justify the expenditure of several hundred thousand dollars. Cost-beneficial SAMAs would most likely be limited to procedure changes and minimal hardware changes. Averted onsite costs (AOSC) is a critical factor in cost-benefit analyses and tends to make preventive SAMAs more attractive than mitigative SAMAs (which improve containment performance but do not impact CDF).

Table 3 shows the average and ranges of CDF, population dose, $/event, $/person-Sv [$/person-rem], and MAB computed for all approved U.S. license renewals as of August 2009. Figure 1 shows typical cost benefit thresholds for different reductions in CDF per year and person-Sv per year, assuming a 3% discount rate, a 20 year term, and other cost factors provided in NUREG/BR-0184 and NUREG/BR-0058.

Table 3. CDF, Population Dose, and Maximum Attainable Benefit Associated with Completely Eliminating Severe Accidents

	Average	Range
CDF (/yr)	4.0×10^{-5}	$1.9 \times 10^{-6} – 3.3 \times 10^{-4}$
Population Dose (person-Sv/year [person-rem/year])	0.15 [15]	0.006 – 0.69 [0.6 – 69]
$/event	$2.8 billion	$49 million – $12 billion
$/person-Sv [$/person-rem]	$220,000 [$2,200]	$69,000 - $670,000 [$690 – $6,700]
Total MAB	$1.7 million	$110K – $8.7 million

Figure 1 . Typical Cost Benefit Threshold (3% Discount, 20 Year Term)



The SAMA identification and evaluation process has matured over the years, and typical analyses for nuclear power plant license renewal are now identifying multiple cost-beneficial SAMAs for most plants.

6. Potentially cost-beneficial SAMAs

Numerous potentially cost-beneficial SAMAs have been identified to date in U.S. operating nuclear power plant license renewal applications that have been approved. Most of these SAMAs are low-cost improvements such as modifications to plant procedures or training, minimal hardware changes to enable cross-tying existing pipes or electrical buses, and using portable equipment (e.g., generators and pumps) as backups. Below we provide examples of the specific potentially cost-beneficial SAMAs that have been identified for different operating U.S. plants.

SAMAs related to station blackout or loss of power sequences:

- Use portable generator or portable battery charger to extend coping time in loss of AC power events, or extend DC power availability
- Procure an additional portable 480VAC station diesel generator for backup to EDGs
- Install minimal hardware modifications and modify procedures to provide cross-tie capability between 4 kv AC emergency buses
- Modify plant procedures to allow use of a portable power supply for battery chargers, which would improve the availability of the DC power system
- Use the security diesel generator to extend the life of the 125 VDC batteries
- Modify plant procedures to use DC bus cross-ties to enhance the reliability of the DC power system
- Install key-locked control switches to enable AC bus cross-ties
- Develop procedures and operator training for cross-tying an opposite unit diesel generator

SAMAs related to internal floods, fire, seismic, and other external events:

- For internal floods, install watertight door or watertight wall around vulnerable equipment
- Install interlocks to open doors on high water level in order to divert flood water to a safe area, and change door swing direction to prevent opening a flood path to battery rooms
- Waterproof motor operators for vulnerable valves to mitigate floods caused by service water line breaks
- Enhance protection of critical fire targets by improving separation or providing cable tray protection
- Modify RHR valve yokes to reduce risk from seismically-induced ISLOCA
- Provide additional diesel fire pump for fire service water system (develop procedure for the use of a fire truck to pressurize and provide flow to the fire main)
- Increase fire pump house building integrity and combustion turbine building integrity to withstand higher winds, so that fire system and combustion turbines would be capable of withstanding a severe weather event

SAMAs related to protection systems:

- Change logic in under-voltage, block, and/or actuation signals, e.g., to 3 out of 4 logic
- Modify procedures to allow operators to defeat the low reactor pressure interlock circuitry that inhibits opening the LPCI or core spray injection valves following sensor or logic failures that prevent all low pressure injection valves from opening
- Install additional fuses in control panel to enable direct torus vent valve function during loss of containment heat removal accident sequences

SAMAs related to support systems:

- Various SAMAs to improve cooling of EDG rooms, e.g., revise operator procedure to provide additional space cooling to the EDG room via the use of portable equipment; modify plant procedures to open the doors of the EDG building upon receipt of a high temperature alarm; install diverse fan actuation logic for starting EDG room fans or operating exhaust dampers
- Provide an alternate/additional compressor that can be aligned to the instrument air supply header

SAMAs related to procedures and training:

- Increase operator training on the systems and operator actions determined to be important from the PRA
- Modify procedures and training to operate the isolation condensers with no support systems available
- Develop guidance/procedures for local, manual control of RCIC following loss of DC power
- Emphasize timely recirculation swap-over in operator training
- Develop emergency procedures for refilling the condensate storage tank using the fire service water system
- Use firewater systems as backup for containment spray
- Develop procedure for local manual operation of AFW when control power is lost

7. Conclusion

PRA has been used to identify cost-beneficial improvements at numerous operating U.S. nuclear power plants. Importance measures are used to identify risk-significant basic events from the PRA, and SAMAs are identified to address these basic events. SAMAs that are found to be potentially cost-beneficial tend to be low-cost improvements such as modifications to plant procedures or training, minimal hardware changes, and use of portable equipment.